Licensing Modernization Project (LMP)

Molten Salt Reactor Experiment (MSRE) Case Study
Using Risk-Informed, Performance-Based Technical Guidance to Inform Future Licensing for Advanced Non-Light Water Reactors

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MOLTEN SALT REACTOR EXPERIMENT (MSRE) CASE STUDY

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Acronyms and Abbreviations

ACRS: Advisory Committee on Reactor Safeguards

AEC: U.S. Atomic Energy Commission

AOO: Anticipated Operational Occurrence

BDBE: Beyond Design Basis Event

CCS: Component Cooling System

DBA: Design Basis Accident

DBE: Design Basis Event

DID: Defense-in-Depth

DOE: U.S. Department of Energy

EAB: Exclusion Area Boundary

EPRI: Electric Power Research Institute

ET: Event Tree

ETA: Event Tree Analysis

F-C: Frequency-Consequence [target]

FT: Fault Tree

FTA: Fault Tree Analysis

HAZOP: Hazards and Operability [study]

IDP: Integrated Decision-Making Panel

IE: Initiating Event

LBE: Licensing Basis Event

LMP: Licensing Modernization Project

MCA: Maximum Credible Accident

MHTGR: Modular High Temperature Gas Reactor

MSR: Molten Salt Reactor

MSRE: Molten Salt Reactor Experiment [specific MSR design]

NEI: Nuclear Energy Institute

Non-LWR: Non-Light Water Reactor

NRC: U.S. Nuclear Regulatory Commission

OGS: Off-Gas System

ORNL: Oak Ridge National Laboratory

ORO: Oak Ridge Operations

PHA: Process Hazards Analysis

PRA: Probabilistic Risk Assessment

PrHA: Preliminary Hazards Analysis

PSF: PRA Safety Function

RG: Regulatory Guide

RIPB: Risk-Informed and Performance-Based

RORC: Reactor Operations Review Committee

RR: Residual Risk

RSF: Required Safety Function

SAR: Safety Analysis Report

SR-SSC: Safety Related Systems, Structures, and Components

SSC: Systems, Structures, and Components

TRL: Technology Readiness Level

VU: Vanderbilt University

1. Introduction

The Licensing Modernization Project (LMP) is an ongoing industry-driven effort led by Southern Company Services (SCS) with the goal of developing a modernized framework for the licensing of advanced non-light water nuclear reactors (non-LWRs). The LMP initiative is intended to develop a technology-inclusive, risk-informed, performance-based (RIPB) process that can be applied for licensing the next generation of non-light water reactors. A key element of the LMP is the development of associated guidelines for establishing, evaluating, confirming, and documenting the adequacy of defense-in-depth (DID) for advanced non-light water reactor technologies. This effort is focused on non-LWR technologies, because of the need for transparent and principled guidance for the advancement of the multiplicity of novel designs that are being developed. However, given the significantly reduced risks associated with many of the advanced reactor designs as compared to LWRs, a non-risk-informed application of DID could result in excess conservatism.

In parallel, EPRI has worked with Vanderbilt University to conduct an independent project under its Advanced Reactor Strategic Program to define an approach that facilitates the design-to-license process via the integration of safety analysis elements from established methods of Process Hazard Analysis (PHA) and Probabilistic Risk Assessment (PRA). The second phase of this EPRI project seeks to demonstrate the resulting PHA-to-PRA Methodology through one or more case studies using well-characterized specific systems or sub-systems drawn from non-LWR designs. Given the complementary nature of activities in both LMP and EPRI's PHA-to-PRA project and the mutual value of a well-developed case study, Southern Company Services and EPRI executed a joint project to coordinate and leverage the two efforts.

This letter report documents the application of EPRI's PHA-to-PRA methodology as part of the RIPB LMP process described in NEI 18-04 [1], which offers one approach for licensing of non-LWRs. The Molten Salt Reactor Experiment (MSRE) was chosen as a reference MSR reactor design given the wealth of accessible, non-proprietary, and publicly available information on design and operational experience.

1.1. Purpose

The resulting MSRE Case Study documented in this letter report is intended to exercise the EPRI PHA-to-PRA methodology and illustrate key LMP processes as described in the LMP Guidance Document, NEI 18-04 [1]. The LMP process has been applied by advanced reactor developers and vendors with generally positive results. The MSRE Case Study builds upon the demonstration projects performed with X-energy [2] and General Electric Hitachi [3].

Following the issuance of the LMP Guidance Document as NEI 18-04 in September 2018, staff from the U.S. Nuclear Regulatory Commission (NRC) and industry representatives briefed the Advisory Committee on Reactor Safety (ACRS) Future Plant Subcommittee in June and October 2018 and the full ACRS in February 2019. Subsequently, the NRC published draft regulatory guide DG-1353, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," in the Federal Register on May 3, 2019 for public comment [4]. This proposed regulatory guide endorses, with clarifications, the principles and methodology in NEI 18-04 as one acceptable method for determining the appropriate scope and level of detail for parts of applications for licenses, certifications, and approvals for non-LWRs.

For the MSRE Case Study, the LMP approach is applied to a specific inventory of radionuclides in a liquid-fueled Molten Salt Reactor (MSR) design. MSRs are an example class of advanced reactor designs that lacks a significant body of PRA case studies due to the relatively immature stage of development. The work described in this report evaluated the MSRE design at an assumed pre-operational stage of design[1] in an attempt to demonstrate how a reactor designer could employ hazard analysis tools and the LMP approach to develop RIPB insights early in the design process.[2]

For the simulated early design stage, the MSRE design and engineering reports (including Refs. [7-9]) were used to provide the information typically represented in system description documentation, while Oak Ridge National Laboratory (ORNL) MSRE documentation discussing actual system and component design performance (such as [10]) served as a proxy for the design team that would interface with the safety analysis team members during PHA studies and the subsequent development of Fault Tree Analysis (FTA) and Event Tree Analysis (ETA). Analyzing the MSRE at this level of design maturity allowed the MSRE Case Study team to investigate how a designer might use tools such as PHA studies to initiate the first iteration of the RIPB-based LMP process. This concept is discussed on page 9 of DG 1353, which states the following [4]:

> *The incorporation of safety analysis methods appropriate to early stages of design, such as FMEA and PHA, provide industry-standardized practices to ensure that such early stage evaluations are systematic, reproducible and as complete as the current stage of design permits. The subsequent use of the PRA to develop or confirm the events, safety functions, key SSCs, and adequacy of defense in depth provides a structured framework to risk-inform the application for the specific reactor design.*

Finally, this report provides observations noted by the EPRI-VU team after exercising the LMP approach during the assessment.

[1] i.e., a Technology Readiness Level (TRL) of ~4-5. For more information on the concept of TRLs, see Refs. [5] and [6]

[2] One exception to this approach is the use of measured radionuclide concentrations for the consequence calculations below. This operational data was used because the pre-operational calculations are contained in ORNL documents (ORNL CF-57-7-17 and ORNL-MSR-61-101) that are currently in the process of review for release and were not available for use in this work; it is intended that they will be used to update the results, if made available.

1.2. Scope

As discussed in EPRI Report 3002011801 [11], a starting point for developing a model to analyze risk in a reactor design can be the performance of a PHA study using one or more industry-standard methods. Detailed discussion of PHA and the MSRE design are available in Refs. [12-14]. The following paragraphs discuss the process and results of defining and selecting which nodes of the MSRE were of highest priority for quantitative risk assessment, along with the subsequent LMP assessments that were performed.

In order to conduct a PHA, it is necessary to divide the reactor design into analyzable sections or "nodes." Based on a review of MSRE design information [7-9], 21 relevant nodes were identified based on primary function and nominal operating conditions. The list of nodes includes:

- Fuel salt loop
- Fuel salt drain/fill system
- Fuel salt processing equipment
- Coolant salt loop
- Coolant salt drain/fill system
- Sampler-enricher system
- Cover gas system
- Leak detection system
- Fuel salt off-gas system
- Coolant salt off-gas system
- Containment ventilation system
- Component cooling system
- Secondary component cooling system
- Instrument air system
- Treated cooling water system
- Tower cooling water system
- Vapor condensing system
- Liquid waste system
- Drain tank afterheat removal system
- Salt pump lube oil system
- Electrical system

PRA models are typically developed for a specific combination of radionuclide source, plant operating state, and hazard group [15]. As discussed further in Ref. [16], another important step of system characterization was to develop an understanding of which nodes might have significant inventories of radiological materials. Performing a PHA on nodes that contain, or interface with, significant inventories of radioactive material will help identify event sequences that could have associated potential consequences of interest to designers and regulators. Early identification and characterization of radionuclide sources is also consistent with the PRA development process suggested for advanced reactors by the LMP approach [1].

Some of the nodes identified in the MSRE do not differ substantially from systems with significant industrial experience (e.g., tower cooling water system, instrument air system) and others of the nodes may not be common to modern commercial MSR designs (e.g., the sampler-enricher). Additionally, nodes with greater inventories of radioactive material are of more immediate interest because the consequences of event sequences associated with these nodes have the potential to be more severe than those associated with other nodes. Thus, the first MSRE nodes selected to be analyzed in a Hazards and Operability (HAZOP) study were the off-gas system (OGS), the component cooling system (CCS), the fuel salt processing equipment, and the fuel salt loop.

The function of the MSRE OGS is to separate and store volatile activation and fission products from the fuel salt. Additionally, because the effluent of the OGS charcoal beds is exhausted to the MSRE stack, the OGS has an interface with the environment. This function, and form of radionuclides, is unique to MSRs; thus, event sequences associated the radioactive material in this system were the focus of the MSRE Case Study.

Although the CCS does not contain a significant radioactive material inventory during normal operations, the system: performs functions that will likely need to be addressed in most or all MSR designs, was integral to safe operation of the MSRE, and has not been the subject of detailed prior hazard evaluations or risk assessments. Also, in the MSRE design, the CCS interfaces with the reactor cell atmosphere, which can become contaminated if radionuclides from the fuel salt loop or OGS are transported past the first barrier to their release. The MSRE CCS also has a direct interface with the MSRE stack and the environment and is important to understanding the function of the OGS.

There are limitations on the effectiveness of the MSRE design as a proxy for contemporary commercial MSR design concepts. Because the MSRE was licensed as a non-power reactor on a national laboratory site in the 1960s, the approach to containment/confinement may not apply directly to the modern MSR designs. These differences could change the hazards and accident analyses, along with the particulars of any postulated release. **Therefore, the results of the simplified quantitative calculations done in this report for the MSRE should not be considered to constitute a baseline or reference point for modern MSR advanced nuclear power concepts.**

1.3. Objectives

The objective of the MSRE Case Study was to jointly demonstrate the EPRI PHA-to-PRA methodology [11] and implement the LMP approach described NEI 18-04 to illustrate how an MSR design team could early in the design process: evaluate Required Safety Functions (RSFs); identify Systems, Structures, and Components (SSCs) that would make good candidates to be classified as Safety Related SSCs (SR-SSCs); and investigate Defense-in-Depth (DID). As noted above, the reactor design being used to demonstrate this process is the MSRE, and more specifically, the analysis in this report will evaluate event sequences that involve the inventory of radioactive material in the OGS. The MSRE design was chosen because of the wealth of publicly available design and operations literature. The OGS radioactive material inventory was selected for evaluation because there is little existing literature that addresses the quantitative evaluation of risk associated with an MSR OGS.

It is important to note that the event sequences evaluated in this Case Study have not previously been quantitatively evaluated; consequently, no thoroughly vetted source term calculation exists for the purposes of estimating dose consequences associated with event sequences involving the radioactive material in the OGS of the MSRE. For this reason, preliminary calculations have been made to approximate the dose consequences that result from these sequences, and these calculation results were used to demonstrate key processes described in NEI 18-04. However, it is important to note that these quantitative results contain significant and unquantifiable uncertainty and a reactor developer would likely use these early results to identify and prioritize technical uncertainties that need to be the subject of later tests and/or experiments before proceeding with more rigorous analysis.

The frequency and consequence estimates developed during this Tabletop Exercise were used to relate the OGS Licensing Basis Events (LBEs) to the frequency-consequence (F-C) target curve suggested in NEI 18-04. This enables the results to provide a demonstration of how to investigate the identification of RSFs for the OGS LBEs and explore how an early stage reactor developer might identify SSCs that are candidates to be classified as SR-SSCs. Figure 4-1 of NEI 18-04 and the associated discussion in Section 4 provided the major guidance for this Tabletop Exercise. Finally, Figure 5-3 and Table 5-2 of NEI 18-04 were used to identify the functions and SSCs that contribute to each layer of DID for the inventory of radionuclides in the OGS, based on the quantitative Event Tree (ET) model that was developed. The results of the PHA studies on the MSRE design using the HAZOP approach [17, 18] were also used to contribute to these investigations.

In contrast with the other LMP demonstrations [2, 3], the MSRE Tabletop team did not have the ability to directly interface with the MSRE design team. As discussed in detail below, uncertainties were encountered regarding data to develop an estimate of source term. Without the ability to conduct experiments to gather additional data to support the dose consequence calculations, assumptions and simplifications had to be made in order to provide quantitative results that could be used to demonstrate the key LMP processes described in NEI 18-04. Therefore, within this MSRE Case Study letter report, emphasis was placed on the objective to demonstrate these processes, rather than on further refining the quantitative frequency and consequence results that were developed.

1.4. Prior Deliverables

Prior deliverables and other recent materials related to this MSRE Case Study include the following:

- A 2017 workshop (and the resulting proceedings) that focused on early integration of safety analysis methods and tools in the reactor design process to guide, optimize, and prioritize design activities [19]

- A 2017 letter report detailing the initial OGS and CCS HAZOP studies and containing the tabular HAZOP study results [17]
- A presentation to NRC staff discussing early progress and results related to the MSRE Case Study on June 15, 2018 [20]
- A 2018 ORNL Technical Report exploring the concept of applying a RIPB LBE selection process to the MSRE based on original ORNL safety analyses [16]
- A 2018 letter report discussing the results of preliminary HAZOP studies on 4 select nodes from the MSRE design [18]
- A 2018 letter report discussing the development of initial ET and Fault Tree (FT) models based upon the HAZOP results [21]
- A 2019 letter report discussing the quantification of frequency and consequence estimates for MSRE OGS LBEs and the use of these results to support further LMP analyses [22]
- A presentation to NRC staff discussing the topics in the above letter report on June 5, 2019 [23]
- An upcoming 2019 Electric Research Power Institute (EPRI) technical report compiling all MSRE related hazard and risk analysis performed to date, including information covered in the letter and technical reports listed above and the conference papers discussed below

2. Background

The development of a RIPB licensing framework has been frequently identified as an important enabler for the commercialization of advanced reactor designs. The LMP's objective is to develop technology-inclusive, RIPB regulatory guidance for licensing non-LWRs for the NRC's consideration and possible endorsement. The NRC has engaged with the LMP team and the Nuclear Energy Institute (NEI). The MSRE Case Study illustrates the application of the LMP RIPB processes and, as noted earlier, has been briefed to NRC staff on two occasions [20, 23].

Following NRC staff review of predecessor white papers and a number of NRC public meetings, [3] the industry issued its consolidated LMP Guidance Document (NEI 18-04 [1]), which describes an approach for use by reactor developers to select LBEs; classify SSCs; determine special treatments and programmatic controls; and assess the adequacy of a design in terms of providing layers of DID. The NRC published draft regulatory guide DG-1353, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors," in the Federal Register on May 3, 2019, for public comment.

[3] https://www.nrc.gov/reactors/new-reactors/advanced.html

3. Demonstration Overview

3.1. Summary of Demonstration Activities

Beginning with the original MSRE design information, the inventories of radioactivity in the MSRE design were identified and characterized [16]. This characterization was used to down-select the radioactive material inventory in the MSRE OGS as the focus of this Case Study. The OGS and the CCS were then analyzed in a HAZOP study. Using the results of these HAZOP studies, ETs, and FTs were structured and evaluated for events of interest pertaining to the OGS radioactivity. One initiating event (IE) was selected and an ET was constructed, using the HAZOP results and the original MSRE systems information, to identify the pivotal events that structure the event tree and discern between individual event sequences. Each pivotal event was then represented by a FT model, the structure of which was based upon the HAZOP results and available MSRE systems information.

Using these ET and FT models, the frequencies were estimated for LBEs that begin with the IE of a rupture of the OGS line immediately downstream of the fuel salt pump bowl. Uncertainties in the frequency estimates were addressed in the models and characterized using EPRI's CAFTA Software [24]. Once the OGS event sequences had been classified based on the frequency bins suggested in NEI 18-04, the next step was to quantitatively estimate the dose consequences associated with each event sequence that had been modeled in the preliminary ETA. Although there is a lack of detailed information available to accurately characterize the composition and behavior of the radioactive material in the OGS flow, a total dose at the Exclusion Area Boundary (EAB) was estimated, as called for in the LMP process. The EAB for the MSRE is defined as 3000m from the MSRE building [25] and a preliminary dose consequence was calculated for an assumed maximally exposed individual.

Using the LBEs identified, the application of the LMP approach was explored to illustrate how an MSR developer at an early stage of design might initiate the first iteration of tasks such as identification of RSFs, classification of SR-SSCs, and evaluation of DID. The work summarized below is discussed in detail in four EPRI letter reports [17, 18, 21, 22].

3.2. Demonstration Prerequisites and Inputs

The only MSR focused on civilian applications that has been operated is the MSRE.[4] Because authorization to construct and operate the MSRE was issued by the Atomic Energy Commission (AEC) in the 1960s, a PRA model was not required and thus was not developed for the MSRE. To date, there is no publicly available PRA model for an MSR design. The following subsections describe the context for the MSRE safety case, the original ORNL MSRE documentation referenced , and relevant conference papers.

[4] The Aircraft Reactor Experiment (ARE) was operated in November 1954 at ORNL with a NaF-ZrF4-UF4 fuel for a total thermal power production of 96 MW-hr; however, this sodium cooled reactor was designed to investigate aircraft nuclear propulsion rather than ground-based civilian power production [26].

3.2.1. Summary of Initial MSRE Safety Case[5]

The U.S. Atomic Energy Commission licensed a wide variety of non-LWRs mostly using customized safety reviews based on engineering experience and judgement. Since the late 1960s, non-LWRs have typically been licensed using the same process as the LWRs with the use of exceptions and exemptions where LWR requirements were considered inadequate or did not apply. Before the late 1960s, licensing for reactors began with a Preliminary Hazards Analysis (PrHA). This PrHA formed the basis for the safety analysis and was reviewed by the AEC Regulatory Division for the commercial licensing. Experimental and test reactors followed a less prescriptive pathway. The PrHA and Safety Analysis Report (SAR) were reviewed by the ACRS. In the SAR, the applicant proposed a postulated Maximum Credible[6] Accident (MCA), which was believed to bound all other accident consequences and was used to estimate offsite consequences. Thus, the MCA was used in a manner similar to what would later be called a "design basis accident"[7] in the safety analysis and system design processes.

Regarding the licensing of experimental, research, and test reactors at ORNL, the reactor design was reviewed by an independent group of experts from various disciplines within ORNL reporting to the Laboratory Director, known as the Reactor Operations Review Committee (RORC). Reviews began with conceptual design and continued through construction and operation. The PrHA and Safety Analysis documentation were prepared by the ORNL project team and reviewed by the RORC. ORNL also usually set up an independent review committee of outside consultants. Results of the ORNL review of the PrHA and SAR, along with the actual documents, were then presented to the AEC Oak Ridge Operations (ORO) office for review and comment.

The ORO Review results, along with the ORNL documents, were sent to the program office at AEC Headquarters. In the case of the MSRE, the Reactor Technology Development Office was the program office. AEC Headquarters was the authorizing organization; the PHA and SAR were reviewed by the program office and the ACRS. If approved, these groups made a recommendation to the AEC and the AEC then could grant a construction permit for the reactor. The process was then repeated for the MSRE operating license. An operational readiness review was conducted by ORO and the project office at AEC headquarters prior to startup.

The PrHA and the SAR for the MSRE were based on previous ORNL experience with fluid-fueled reactors, more specifically the Aircraft Nuclear Reactor Experiment and two aqueous homogeneous reactor experiments. The MSRE PrHA was first published in 1960 and reissued in 1962 as the "Molten Salt Reactor Experiment Preliminary Hazards Report", ORNL CF-61-2-46 Addendum No. 2 [28]. Supporting safety analyses were published in "Safety Calculations for MSRE" [29].

[5] This section draws from a presentation made by George Flanagan at the 2017 ORNL MSR Workshop, entitled "Historical Perspective – MSRE Safety Basis Authorization" [27].

[6] "Credible" meaning useful as a bounding analysis, but not necessarily mechanistic.

[7] Accidents that form the design basis for engineered safety features.

The MSRE PrHA was basically a barrier analysis, as the safety approach was that each area of the MSRE that contained fuel or fission products was surrounded by at least 2 barriers. In the PrHA, events were identified which could partially damage a single barrier but produce no release if second barrier remained undamaged. Events that might damage two barriers were considered to be very unlikely to occur but were thought to contribute to off-site consequences (today these might be considered Beyond Design Basis Events). Identified events were then analyzed as part of the safety analysis to provide detailed information regarding impacts on fuel/graphite and vessel temperature, power, and pressure.

In the MSRE SAR [25], six reactivity events were analyzed using analog computers and an early digital computer code (MURGATROYD, later ZORCH). These events were the following:

- Fuel Pump Failure
- Cold Slug Accident
- Filling Accident
- Loss of Graphite from the Core (filling the empty space with fuel)
- Fuel anomalies (precipitated fuel circulating in core or non-mixed fuel lumps circulating in core)
- Uncontrolled Control Rod Withdrawal

The MSRE team also looked at several ramp/step reactivity additions. The results largely indicated that the consequences associated with these events were benign [25]. Some internal damage to the reactor from high temperatures could result from three events: large cold slug accidents, premature criticality during refueling, and uncontrolled withdrawal of control rods. However, these events could only result from compound failure of protective devices. In each case there existed effective corrective actions, independent of the credited safety function, so damage was considered to be unlikely.

In addition to reactivity events, the final SAR of the MSRE investigated the following:

- Loss of Flow
- Loss of Heat Sink
- Decay Heat Removal
- Criticality in Drain Tanks
- Freeze valve and flange failures
- Excessive wall temperatures
- Corrosion
- Salt spillage
- Beryllium release from a leak

The SAR [25] describes the most probable accident for the MSRE as a small leak of fuel salt into the secondary container.[8] In this scenario, radiation monitors were intended to shut down the reactor and alarm the operators. Airborne activity that had been released could then be pumped from secondary containment and through a charcoal bed and filters before being released up the stack. The calculations in the SAR indicated that the dose consequences associated with this scenario did not exceed maximum permissible dose on-site at ORNL.

The MCA for the MSRE was considered to be a break in the 1.5-inch drain line or a break in a 5-inch fuel line. The calculations in the SAR assumed that the amount of salt released from both lines would total 10,000 lbs (4000 from fuel and 6000 from drain line). The calculations also assumed a simultaneous spillage of water into the secondary containment to maximize pressure in the cell (calculated to be 110 psig without venting). The rupture disk in the vapor condensing system was designed to open at 20 psig, and the calculations estimated the maximum pressure in the reactor cell would be 39 psig in this case. A cell pressure of 39 psig was not considered to be high enough to cause the cell to fail. Assuming 1% leakage at 39 psig, the dose offsite (3000 m) under the worst meteorological conditions was calculated to be 6 rem from iodine [25].

In order to arrive at these numbers, the SAR reports that the calculations assumed that 10% of the total inventory of iodine[9], 10% of the particulates, and 100% of the noble gases were released. The bases for these assumptions have not been confirmed as part of this work, and there is no analysis of any possible accidents involving the radioactive material handled by the OGS in either the MSRE PrHA or SAR. As previously mentioned, this means that a thoroughly vetted source term calculation does not exist for the purposes of estimating dose consequences associated with the material in the MSRE OGS. The behavior of the iodine is noted as a key area of the analysis uncertainty that would need to be resolved for subsequent iterations of the LMP process as the safety analyses mature.

3.2.2. Summary of MSRE Technical Documents

This subsection contains a brief overview of the data sources used to facilitate the MSRE Case Study.

Reports written before operation of the MSRE:[10]

- Part I, Description of Reactor Design (ORNL-TM-728) [7]: This report was written to discuss the design of the reactor. It contains thorough discussion of the design details and it contains the original design flowsheets for each MSRE subsystem.

[8] For the fuel salt loop, the secondary container is the reactor cell.

[9] The calculations in the SAR also assume that 50% of the iodine released subsequently plates out on secondary container surfaces and states, "based on experiments in which the solubility of the fuel salt in water was measured, much less iodine is expected to be released." [25]

[10] Each of the reports in this list are a volume in a larger report and their title begins with the words "MSRE Design and Operations Report," which has been omitted here for brevity.

- Part IIA, Nuclear and Process Instrumentation (ORNL-TM-729A) [8]: This report details the design and intended function of the MSRE safety system (i.e., most of the automatic responses of the system).
- Part III, Nuclear Analysis (ORNL-TM-730) [30]: This report discusses the calculations made to characterize the nuclear behavior of the MSRE.
- Part VI, Operating Safety Limits for the MSRE (ORNL-TM-733) [31]: This report describes the operating safety limits for the MSRE that are intended to protect the safety and health of the public, the safety of the operators, and the safety of the system against a severe and disabling accident.
- Part VIII, Operating Procedures (ORNL-TM-908, Vol. II) [32]: This report contains the written operating procedures for the system. Volume I of ORNL-TM-908 discusses the nuclear aspects of operation, the operation of auxiliary systems, and startup checklists for the auxiliary systems.

<u>Reports written after the conclusion of MSRE operations:</u>

- Fission Product Behavior in the MSRE (ORNL-4865) [33]: This report contains much of the data taken during and after MSRE operations regarding the behavior of fission products in the system, as well as interpretation of the data.
- MSRE Systems and Components Performance (ORNL-TM-3039) [10]: This report contains some high-level schematics for the final configurations of some systems and a detailed discussion of MSRE operating experience.
- MSRE Design and Operations Report – Part IIB, Nuclear and Process Instrumentation (ORNL-TM-729B) [9]: This report contains a thorough discussion of the instrumentation for almost all of the MSRE systems, as well as detailed drawings documenting the final configuration of each system.

3.2.3. Summary of Prior Conference Papers

The following conference papers cover topics related to the MSRE Case Study and discuss portions of the work that has been conducted to date:[11]

- 2017 ANS Winter Meeting - "Preliminary Risk Assessment of a Generalized Molten Salt Reactor Off-Gas System" [34]
- International Congress on Advances in Nuclear Power Plants (ICAPP 2018) - "Preliminary Hazard Assessment and Component Reliability Database for the Molten Salt Reactor Experiment" [12]

[11] Full citations are provided in the References section of this report.

- Probabilistic Safety Assessment and Management (PSAM14) - "A Project to Encourage the Early Integration of Safety Assessment into the Design, License, and Build Process of Nuclear Power Plants – Status Report" [14]
- Probabilistic Safety Assessment and Management (PSAM14) - "Application of a Method to Estimate Risk in Advanced Nuclear Reactors: A Case Study on the Molten Salt Reactor Experiment" [13]
- 2018 ANS Winter Meeting - "Preliminary Reliability Analysis of Molten Salt Reactor Experiment Freeze Valves" [35]
- International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 2019) - "Development of a Methodology for Early Integration of Safety Analysis into Advanced Reactor Design" [36]

3.3. LBE Selection

The LBE selection process follows the approach described in Chapter 3 of NEI 18-04 [1]. However, since this Case Study represents only a first iteration of the LBE selection process for the MSRE OGS, it was also necessary to use the concepts displayed in Figure 3-3 of NEI 18-04 to develop building blocks for an MSRE-specific PRA model for the radioactive material in the OGS.

3.3.1. Summary of MSRE OGS Radioactive Material, HAZOP Studies, ETA, and FTA

The following subsections discuss the process of characterizing the radioactive material in the MSRE OGS, the results of the HAZOP studies on the associated systems (OGS, CCS), and the construction of the ET and FT models used to investigate LBEs.

Characterization of Radioactive Material in MSRE OGS

For the LMP approach described in NEI 18-04, PRA model development starts with identifying and characterizing the sources of radioactive material that apply to the PRA's scope. This step is particularly relevant to the analysis of a liquid-fueled MSR. During normal operation of an MSR, the radioactive isotopes and the fission products generated in the fuel are dissolved and circulating within the fuel salt loop. The volume in this salt loop is larger, on a per-kilowatt basis, than that of the structures confining these isotopes in other reactor designs, such as the fuel assemblies of an LWR. Also, due to the chemistry of the fuel salt, the fission products in the MSR system can exist in multiple physical states and chemical compounds under normal operating conditions within the system.

When fresh fuel salt was loaded into the MSRE, the main radioactive constituents were the fertile and fissile isotopes of the actinides. However, during MSRE operation, fission products and activation products were generated in the fuel salt by nuclear reactions. ORNL determined that there were three principle types of fission products in the fluoride-based MSRE fuel salt, distinguished by their specific mechanism(s) of migration [33]. These groups of fission products and their approximate distribution, under normal operating conditions in the MSRE (in weight percent, wt% [37]), were as follows:

1. Salt seekers (e.g., Sr, Y, Zr, I, Cs, Ba, Ce): 59 wt%, soluble in fuel salt and remain with the fuel salt.
2. Noble metals (e.g., Nb, Mo, Ru, Sb, Te): 24 wt%, reduced by the UF_3 in the fuel salt and exist in salt in the metallic state. They migrated to various surfaces—including graphite, Hastelloy N, and gas-liquid interfaces—and adhered to these surfaces.
3. Noble gases (Kr and Xe): 17 wt%, slightly soluble gases in the fluid fuel, and as such, they were readily stripped from the fuel salt and transported to the OGS.

One aspect that could complicate the tracking of these fission products throughout the MSRE system was that certain fission products would transition between groups according to their respective decay chains. For example, ^{137}Xe was stripped out of the fuel salt as a noble gas, so a significant amount of ^{137}Cs (a daughter product of ^{137}Xe) could be found in the OGS rather than in the fuel salt, even though ^{137}Cs is considered a salt seeker.

The volatile fission products, mostly the noble gases Kr and Xe, comprise the principal radionuclide inventory of concern. During normal operating conditions, some concentration of these isotopes would exist dissolved in the fuel salt, but the introduction of helium into the fuel salt in the bowl of the fuel salt pump was designed to strip a significant portion of these noble gases from the fuel salt (~38% [38]). As fuel salt was sprayed out of holes in a distributor in the pump bowl, the stripped noble gases were drawn into the OGS [7]. The OGS included: a piping run to provide hold-up time for the radioactive gases to decay (~two hours), water-cooled beds of activated charcoal to adsorb noble gases, roughing filters and particle filters, and a stack to dilute any radionuclides remaining when the resulting effluent was exhausted to the atmosphere.

In ORNL calculations made before MSRE operation, the radioactive gas (mixed with helium carrier gas) in the OGS was estimated to remove about 280 curies per second (Ci/s) from the pump bowl into the off-gas line [7]. In the charcoal beds, the residence time of Xe was designed to be at least 90 days, and the residence time for Kr was at least 7.5 days. During this time, almost all of the fission product gases decayed to stable elements. However, because some of the daughters in the decay chains of the fission product gases were particulate in nature (e.g., ^{89}Sr, ^{137}Cs, ^{140}Ba), deposits in the charcoal beds, filters, and piping retained these daughters.[12] Therefore, by the time that the gas stream left the charcoal bed, the only radioactive isotopes that were calculated to exist in any significant amount[13] were ^{85}Kr, ^{131m}Xe, and ^{133}Xe [7].

The event sequences analyzed were related to the gas flow immediately downstream of the outlet of the fuel salt pump bowl, because that portion of MSRE OGS contains the gaseous radionuclides that are at their highest concentration. The most comprehensive available data concerning the composition of

[12] For location of the components discussed in this paragraph, see Figure 1.

[13] These radioisotopes were calculated to have activity concentrations on the order of 1E-6 µCi/cc; all other radioisotopes were estimated to have activity concentrations no greater than 1E-8 µCi/cc. See Table 12.1 of Ref. [7] for more detail.

the radioactive gas leaving the gas space of the fuel salt pump bowl is available in Ref. [39]. Section 7.5.1 of that report discusses the result of gamma spectroscopy measurements that were taken on the off-gas line with the MSRE operating at full power. The ORNL authors of that report present the estimated average activity of individual radionuclides in the off-gas line and analyze the results. The calculations are based upon the values displayed in Table 7.4, and the discussion on pages 74-81 of Ref. [39]. It is important to note that the authors of Ref. [39] suggest that these activities contain significant uncertainty and may be a factor of 10 to 50 too high. Possible explanations for this overestimation include errors by the computer program used during the experiment and longer than normal residence time for certain radionuclides (e.g., adsorption or deposition on the OGS piping). However, because Ref. [39] contains the only data identified to-date that attempts to characterize the radioactivity of the MSRE OGS flow, the data from that report was used as the starting point for the dose calculations in this exercise.

As previously mentioned, pre-operational calculations in Ref. [7] estimated ~280 curies/sec in the off-gas flow from the pump, but an estimated radionuclide makeup of this gas stream is not provided in Ref. [7]. This data is contained in ORNL documents (ORNL CF-57-7-17 and ORNL-MSR-61-101) that are currently in the process of external review for release at ORNL and were not available for use in this work. Using the measurements in Ref. [39] as a basis, ~2260 curies were estimated to be flowing into the off-gas line per second from the gas space of the fuel salt pump. We could not identify information to resolve the discrepancy during the time period and thus used the more conservative measured value of Ref. [39]. The estimated radioactivity for the OGS flow in the pipe at the effluent of the gas space of the fuel salt pump bowl is displayed by isotope in Table A-1 of Appendix A.

One radioelement off-gas concentration with a particularly high level of uncertainty involved in the dose calculations is iodine. According to Ref. [33], the gamma spectrometer studies strongly suggest that iodine left the fuel salt via off-gas; however, neither gas samples nor examinations of OGS components were able to support such a loss path. The MSRE Case Study team concluded that “of the order of one-fourth to one-third of the iodine has not been adequately accounted for [33]” in their fission product material balance. Additionally, the chemical form of this iodine is not well-characterized in existing literature. The MSRE team did conclude, based upon spectroscopy studies of the MSRE system following a shutdown, that iodine activity could be detected due to the decay of I-precursors that deposited on surfaces in the system; however, iodine itself remained largely with the fuel salt [39]. Recent thermodynamic calculations to predict the behavior of iodine in liquid-fueled MSRs predict negligible presence of the elemental gases (i.e., I and I_2) as well as for the gaseous IF_x compounds [40].[14]

Thus, for the calculations documented herein, iodine was assumed to be in a form that behaves like a particulate, in the sense that filters could mitigate the amount of iodine that can be released to the stack after it has been released to the reactor cell (see Figure 3 for a schematic of the release path). Consistent with the results of MSRE pre-operational experiments (and the MSRE design team’s approach

[14] Iodine was indicated to be in the form of an iodide and the description of the full system can be achieved considering cations (Li^+, Cs^+) and anions (F^-, I^-) in a reciprocal system [40].

in the SAR) [41], it was also assumed that half of the iodides that are released to the reactor cell plate out before being available to be released via the stack. Efforts to reduce the uncertainty surrounding the iodine dose calculation (as well as the characterization of the OGS in general) are ongoing; however, the preliminary dose estimates were used to demonstrate how to use quantitative frequency and consequence values to investigate RSFs, candidate SR-SSCs, and DID. However, as previously mentioned, the quantitative consequence Case Study results and the associated uncertainty would need to be resolved in subsequent steps in the design and safety analysis process.[15]

MSRE OGS and CCS HAZOP Studies

A simplified flowsheet of major OGS components is displayed in Figure 1. The primary function of the OGS in the MSRE was to handle potentially radioactive gas flows that originated from different areas and/or processes. These can be summarized as three main gas streams. Each of the gas streams is handled by the OGS in a slightly different manner.

- The first source of gas is the continuous discharge of helium containing radioactive fission products that is swept from the fuel salt pump bowl (described above). This gas flow is taken through several volume holdups to allow for decay of the fission products, as well as charcoal beds for krypton and xenon adsorption.
- The second gas flow in the OGS is the reactor and drain tank atmosphere gas (95% nitrogen) that is exhausted through the stack to maintain these cells at sub-atmospheric pressure. This gas stream is pumped into the OGS from the CCS and would only contain significant levels of radioactivity in the case of a leak through a pressure boundary inside one of these cells (e.g., a leak of fuel salt and/or off-gas).
- The final gas stream handled by the OGS includes intermittent, relatively large flows of helium containing significant amounts of radioactive gases and particulates, which are produced during salt transfer operations. This gas flow is taken through a different charcoal bed than the fuel salt pump bowl effluent.

All three gas flows are recombined before passing through banks of high-efficient particulate filters prior to being significantly diluted and exhausted to the atmosphere via the MSRE stack. Radiation monitors were installed in each of these major flow streams, to monitor whether changes are occurring in the make-up of the off-gas streams. Upon sensing high radiation levels, automatic safety system actions closed valves that isolated the off-gas lines to prevent highly radioactive gas from being released to the environment. The radioactivity of the stack exhaust was also measured after flowing through the stack

[15] The following are examples of some questions that might need to be answered in order to reduce the uncertainty: (1) Other than the radionuclides discussed in Ref. [39], what radionuclides exist in the MSRE OGS flow? (2) In what chemical form do the radionuclides in the OGS flow exist? (3) Are the radionuclides in the OGS flow readily transported out of the reactor cell? (4) To what degree do phenomena like filtration in the ceramic filter and plating out in piping affect the transport of radionuclides from the reactor cell atmosphere to the environment?

filters. High radioactivity alarms based on the stack measurements would notify MSRE operators if high levels of radiation were detected in the stack. However, in the case of a stack alarm, operator action would be required to isolate the off-gas lines from the stack [8].

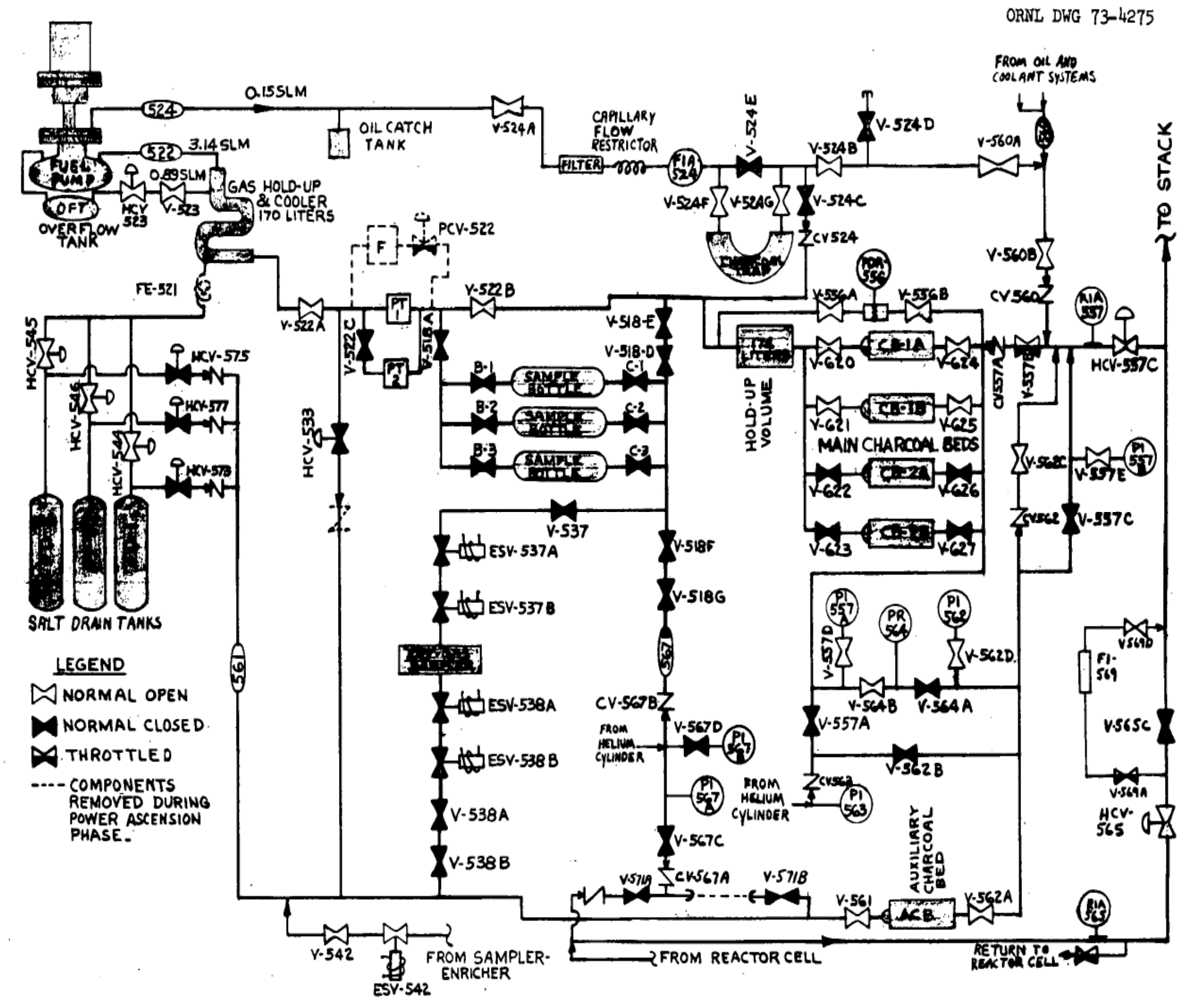

Figure 1: Simplified flowsheet of major OGS components [10]

A primary function of the CCS was to maintain the negative differential pressure in the reactor and drain tank cells (with respect to surrounding areas). Because this system produced the driving force behind this cell evacuation flow, the CCS interfaces with the MSRE OGS (i.e., the second gas flow in the bulleted list above). The CCS was also responsible for cooling freeze valves, which were used to isolate systems and facilitate the movement of fuel salt. Further, the CCS is responsible for the cooling of other components in the reactor cell, including: the first volume holdup for the gaseous effluent of the fuel salt pump bowl and the graphite sampler.

During the HAZOP studies, a total of 35 potential deviations were identified and evaluated for the MSRE OGS [18]. A total of 40 deviations were identified and evaluated for the MSRE CCS. Due to the open nature of the flow circuit in these systems, potential flowpaths for radioactive material to exit the reactor facility were provided. In particular, the presence of volatile fission products in the off-gas stream increases the potential hazard associated with some of the deviations evaluated.

MSRE OGS ETA and FTA

One of the letter reports listed above [21] discusses in detail how the results of the MSRE HAZOP studies, and available component reliability data, were used to structure and evaluate ETs and FTs and estimate the frequency of event sequences pertaining to the inventory of radioactive material handled by the MSRE OGS. Using EPRI's UNCERT software, the uncertainty due to the probability distribution of the component failure rates was calculated. The developed ET model is shown below in Figure 2, and the quantitative frequency estimates are displayed in Table 1.

The IE in the ET model is a leak from one of the off-gas components within the reactor cell, which is the portion of the OGS that contains gaseous radionuclides that are at their highest concentration. Once this radioactive material is released from the OGS to the reactor cell atmosphere, the MSRE system is designed such that this contaminated atmosphere will be drawn into the CCS by the operating component cooling pump blower. A portion of this gas flow is continuously being checked by process radiation monitors, and after passing through the monitors it is exhausted to the atmosphere via the MSRE stack. Similar to the OGS process radiation monitors, if either of the two monitors detects excess radioactivity in the gas flow, multiple automatic system responses are initiated by the MSRE control system. First, a drain of the fuel salt system is initiated. Once the fuel salt has been successfully drained and secured in the drain tanks, the off-gas would be routed to the auxiliary charcoal bed, bypassing the failed portion of the OGS, which would terminate the leak of OGS flow to the reactor cell atmosphere [32].

Secondly, a pneumatic valve downstream of the radiation monitors in the cell exhaust line is intended to close to prevent contaminated cell atmosphere from being exhausted to the stack [8]. Successful isolation of this flow will prevent the radioactive material from being exhausted to the environment. In the case where the fuel salt cannot be drained from the fuel salt loop, but the cell exhaust flow is isolated, operator action would be required to secure the helium cover gas being supplied to the fuel salt pump bowl in order to prevent the possibility of allowing the radioactive effluent gas from the pump bowl to continue leaking into the reactor cell. If this cover gas flow can be secured, the reactor cell would remain at a negative differential pressure, and leakage through the cell would continue to be inward rather than outward.

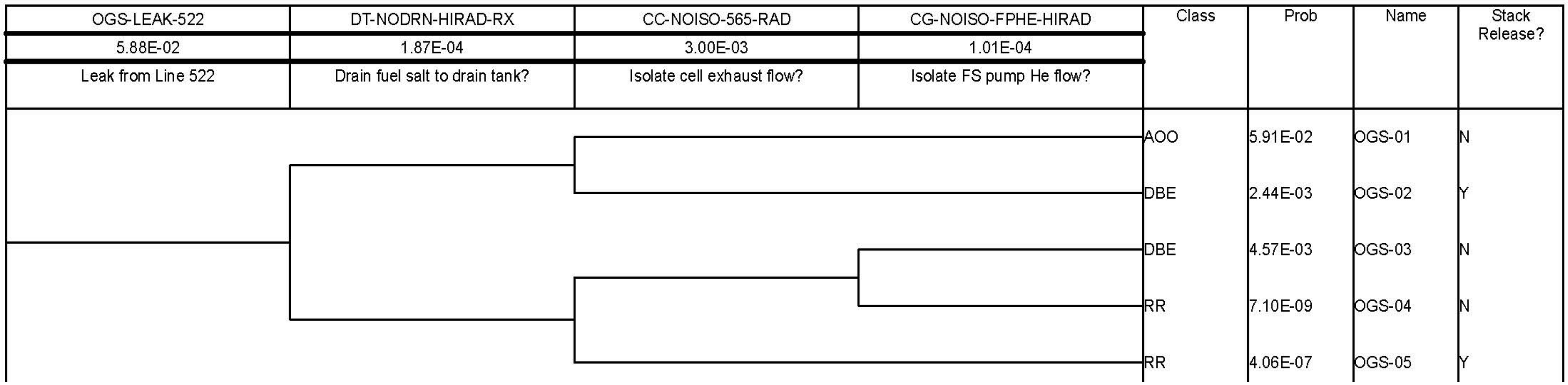

Figure 2: OGS event tree model

Table 1: Summary of event sequences in OGS ETA (N=100,000 for Monte Carlo uncertainty analysis)

Sequence Name	Mean [/reactor-yr]	Classification	Qualitative End-State	Point Estimate [/reactor-yr]	Median [/reactor-yr]	5% [/reactor-yr]	95% [/reactor-yr]
OGS-01	5.91E-02	AOO	Off-gas leak to Rx cell for ~1 hour, stack isolation	5.90E-02	4.30E-02	1.78E-02	1.44E-01
OGS-02	2.44E-03	DBE	Off-gas leak to Rx cell for ~1 hour, release to stack	2.74E-03	8.02E-04	6.66E-05	9.45E-03
OGS-03	4.56E-03	DBE	Off-gas leak to Rx cell for >1 hour, stack isolation, reactor cell negative differential pressure maintained	4.25E-03	1.66E-03	2.41E-04	1.74E-02
OGS-04	7.10E-09	Residual Risk	Off-gas leak to Rx cell for >1 hour, stack isolation, potential to lose reactor cell negative differential pressure	4.49E-09	7.63E-10	3.99E-11	1.99E-08
OGS-05	4.06E-07	Residual Risk	Off-gas leak to Rx cell for >1 hour, release to stack	3.30E-08	4.66E-09	4.27E-10	3.46E-07

Using the event sequence names displayed in Figure 2, a brief summary of each the sequences identified for this release of radioactive material in the OGS is as follows:

- OGS-1 is the scenario in which a leak/rupture occurs in the OGS piping and radioactive off-gas flows from the OGS into the reactor cell. In response to the elevated levels of radioactivity in the cell atmosphere, both a drain of the fuel salt and isolation of the cell exhaust flow are successful.
- For OGS-2, the fuel salt is successfully drained and the leak is terminated, but the cell exhaust flow containing radioactive material is not successfully isolated from the MSRE stack. Similar to the events in OGS-1, the securing of the fuel salt in the drain tank terminates the leak of radioactive material into the reactor cell. Unlike in OGS-1, however, the contaminated atmosphere in the cell has a flowpath to the environment through the CCS and MSRE stack.
- In OGS-3, the cell exhaust flow is able to be secured, but the fuel salt is not successfully drained and secured in the drain tanks. In this scenario, the duration of the OGS leak to the reactor cell is longer than in OGS-1 or OGS-2; however, operator action is able to successfully secure the helium cover gas flow to the fuel salt pump bowl such that the negative differential pressure (monitored and logged parameter) in the reactor cell is maintained.
- OGS-4 is similar to OGS-3 in the sense that the cell exhaust flow is secured, but the fuel salt is not successfully drained; however, in OGS-4 the helium cover gas flow to the fuel salt pump bowl is not secured and the negative differential pressure in the reactor cell would eventually be lost.
- OGS-5 represents the event sequence in which the fuel salt is unable to be drained and the cell exhaust flow is not isolated.

Using the definitions in NEI 18-04, this ET model identifies one Anticipated Operating Occurrence (AOO), two Design Basis Events (DBEs), and no Beyond Design Basis Event (BDBEs). The remaining event sequences are below the frequency threshold for consideration as an LBE; thus, they are considered to be "Residual Risk." [22] It is worth noting that in order to use the frequency definitions for LBEs from NEI 18-04, it was assumed that the MSRE operated at full power for 24 hours a day, 7 days a week, 365 days a year. Because the MSRE was a first-of-a-kind test reactor, both planned maintenance and unplanned shutdowns reduced the availability of the system; thus, estimating a capacity factor of 100% is an overestimation. For example, the MSRE was only critical for about 57% of the elapsed time for the first two years of operation and the percentage is even lower when considering the entirety of its four-year operational period [10].

3.3.2. Allocation of Consequences to the LBEs Identified

In order to compare the events identified in the MSRE case study to the F-C target described in NEI 18-04, a mean dose at the EAB must be estimated for each event sequence. In general, the approach to estimate the dose consequences was to estimate the radioactivity of each radionuclide that is transported through each barrier between the outlet of the fuel salt pump bowl and the environment to calculate the amount of radioactive material that is ultimately released to the atmosphere. As noted below, a comprehensive characterization of radionuclides in the OGS flow is not currently available, but a preliminary attempt to quantify the radioactive material released in each event sequence was made in order to use the results to demonstrate how an MSR developer might use quantitative results in the key processes described in NEI 18-04. Dispersion calculations were then used to estimate the dose at the EAB due to the activity released during each LBE identified for the inventory of radionuclides in the OGS.

It is worth noting that the uncertainties surrounding the calculated preliminary dose estimates are qualitatively high and are not currently quantified. Such uncertainty is a common characteristic of early accident analysis and can be expected to be reduced as the analysis matures to the point where it is consistent with the guidance in the non-LWR PRA standard [15] for mechanistic source terms; however, such an effort was beyond the scope of this Case Study.

The form, distribution, and behavior of key radionuclides, especially iodine isotopes and their precursors, are incompletely described in the literature identified to-date. The preliminary dose consequence values developed in this Case Study do not reflect precise estimates and are intended to be used to demonstrate how the LMP processes may be applied at an early stage in the design process. Consistent with NEI 18-04, any problematic quantitative results would also be used by a developer to inform analysis and design decisions to reduce the frequency and/or consequences associated with a given event sequence.

OGS-1 and OGS-3 Dose Estimates

Figure 3 displays the release path to atmosphere for the radioactivity leaked to the reactor cell from the gas space of the fuel salt pump bowl after the IE of a leak/rupture of the off-gas piping. Referring to the event sequence names in Table 1, in OGS-1 and OGS-3 the flow path to the MSRE stack is secured by the closing of the cell exhaust isolation valve (HCV-565A1) in response to high levels of radioactivity, as can be seen in Figure 3.

During OGS-1, the duration of the leak of radioactive material is assumed to occur for 1 hour. The duration of the leak in OGS-3 is more than one hour; however, in both of these LBEs, there is no immediate driving force to cause the radioactivity to leak out of the reactor cell and into the MSRE building. Although the negative differential pressure between the reactor building and the reactor cell is not able to be permanently maintained with the cell exhaust isolation valve (HCV-565A1) closed, the reactor cell is starting off at a negative differential pressure with respect to the surrounding areas of the building. The reactor cell was designed to have a very low leak rate to the building and tested annually to confirm the low leak rate [10]. Thus, it is assumed that the activity released from the OGS during OGS-1 and OGS-3 would be contained in the reactor cell and the CCS until the pressure of the reactor cell exceeded 0 psig.

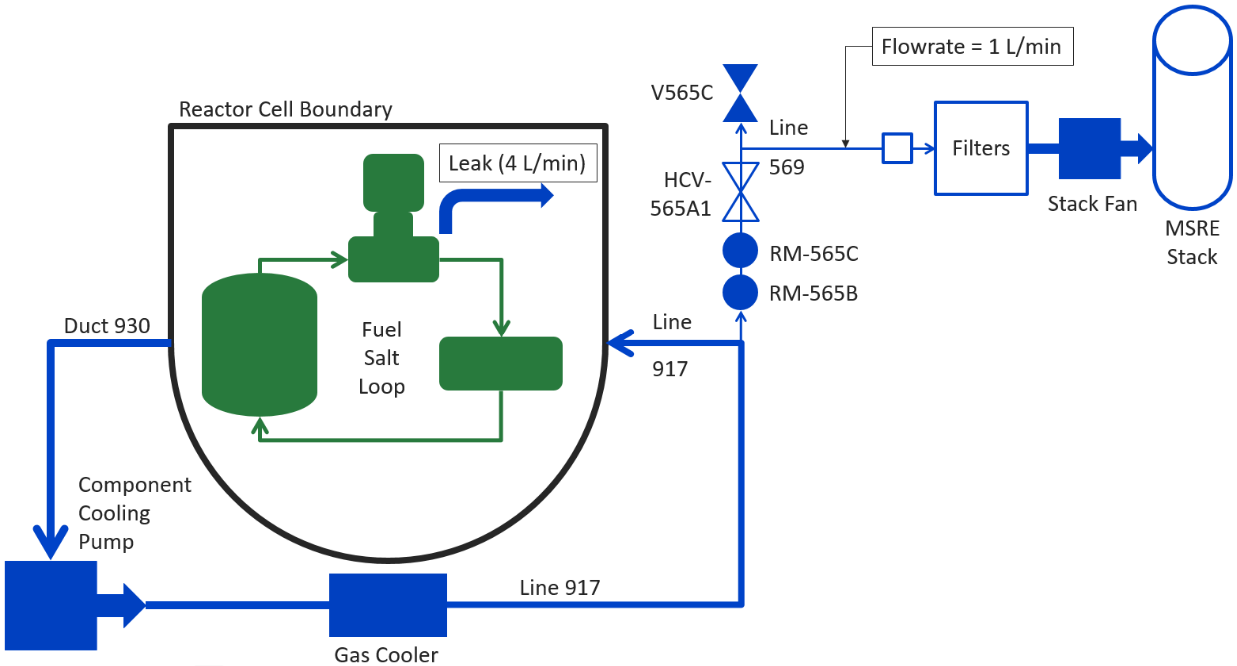

Figure 3: Schematic of barriers and release path of radioactive material for case study LBEs

Calculations indicated that it would take at least 3 days for the reactor cell to reach 0 psig due to the helium cover gas from the fuel salt pump bowl flowing into the reactor cell through the break in the OGS piping and normal inleakage to the cell [22]. Because of the relatively short duration of the OGS leak during OGS-1 compared to the time required to develop a driving force for a leak of the radioactive material out of the cell and the distance between the MSRE building and the EAB, the dose consequence at the EAB for OGS-1 is estimated to be negligible.

Because the fuel salt is not immediately drained in the case of OGS-3, the cover gas from the fuel salt bowl would continue to sweep volatile radionuclides out of the gas space of the fuel salt pump bowl and into the reactor cell through the break in the OGS piping. Although there are no automatic system actions to secure this helium cover gas flow, the MSRE operating procedures [32] would require the operators to secure the flow to the fuel salt pump bowl based on a variety of indications that would be available. In OGS-3, the operators were determined to be able to successfully isolate this flow within 12 hours, which is well within the 3 days required to prevent a loss of the negative differential pressure within the reactor cell. Thus, the dose consequence at the EAB is estimated to be negligible for OGS-3, as well.

OGS-2 Dose Estimate

In OGS-2, the fuel salt is successfully drained and secured in the drain tank after the leak of radioactivity from the off-gas line is detected. After 1 hour, the off-gas has a flow path through the auxiliary charcoal bed in the OGS and does not continue leaking to the reactor cell [32]. As can be seen in Figure 3, the radioactive off-gas would leak from the outlet of the fuel salt pump bowl into the reactor cell atmosphere. The component cooling pump would draw the cell atmosphere (containing the radioactivity) into the CCS. A portion of this flow (about 1 liter/min [7]) is diverted into the cell evacuation line, which would flow through a ceramic filter[16] and then join with the off-gas line from the charcoal beds. The gas would then pass through HEPA filters before being exhausted through the stack. Radiation monitors were designed to send a signal to shut a pneumatic valve upon high levels of radioactivity in the cell exhaust flow, but in OGS-2 it is assumed that this valve remains open.

Per the approach described in NEI 18-04, the dose at the EAB was estimated [22]. The inhalation of particulate fission products was estimated to contribute just over 3.8 rem of inhalation dose at the EAB, and an external dose of 0.164 rem was estimated at the EAB for OGS-2

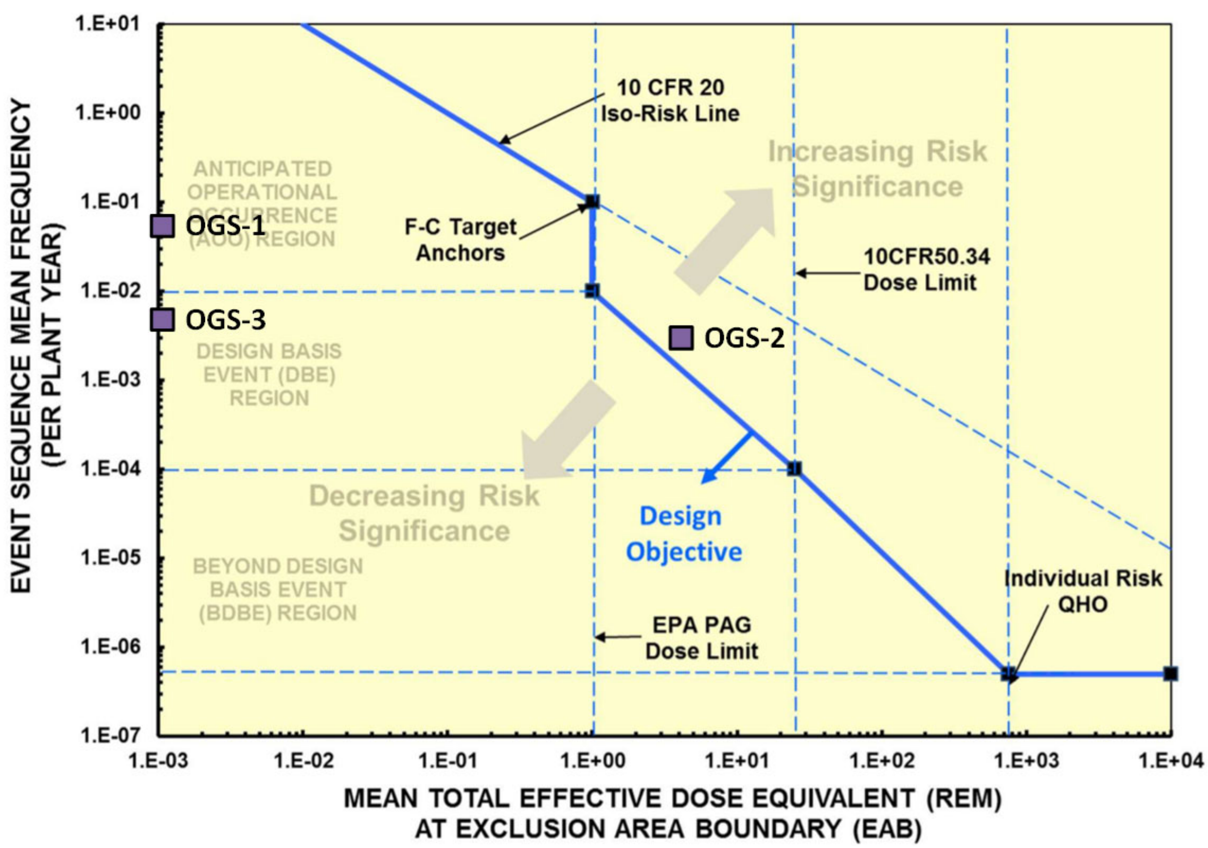

Figure 4: OGS event sequence quantitative results shown on LMP F-C curve

[16] Nominal pore diameter = 60 micrometers

3.3.3 MSRE LBEs Plotted Against the LMP Frequency-Consequence Target

A visualization of the calculated probability and consequences for each analyzed OGS LBE on the suggested LMP F-C curve is shown in Figure 4. The estimates are displayed in the figure as point values; however, uncertainties are present in these preliminary estimates, as previously discussed. The uncertainties in the frequency estimates are quantified and displayed in Table 1.

A consequence estimate has not been completed to-date for OGS-4 or OGS-5, since the estimated frequencies of these event sequences were below the threshold for it to be considered an LBE. Future efforts might involve estimating the dose consequence associated with OGS-4 and OGS-5 in order to investigate for the possibility of cliff edge effects[17]. Regarding OGS-4, there are some uncertainties regarding the exact duration of the off-gas leak to the reactor cell and the point at which the radioactive material in the reactor cell atmosphere is leaked to the reactor building. To calculate a quantitative dose estimate for OGS-5, a more detailed model may need to be developed.

In the process described in NEI 18-04, the quantitative results from LBE analysis are intended to be used for other purposes, including: identifying RSFs, selecting SR-SSCs, and evaluating DID adequacy. Figure 5 illustrates the relationship between these tasks. Although a comprehensive list of LBEs pertaining to the radioactive material in the MSRE OGS has not yet been identified and/or analyzed, the following subsections of this report will explore how an MSR developer at an earlier stage of design could use preliminary quantitative analysis–in combination with qualitative results from a PHA study–to begin the first iteration of these tasks.

[17] A cliff edge effect in a nuclear power plant is an instance of severely abnormal plant behavior caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter, and thus a sudden large variation in plant conditions in response to a small variation in an input. [42]

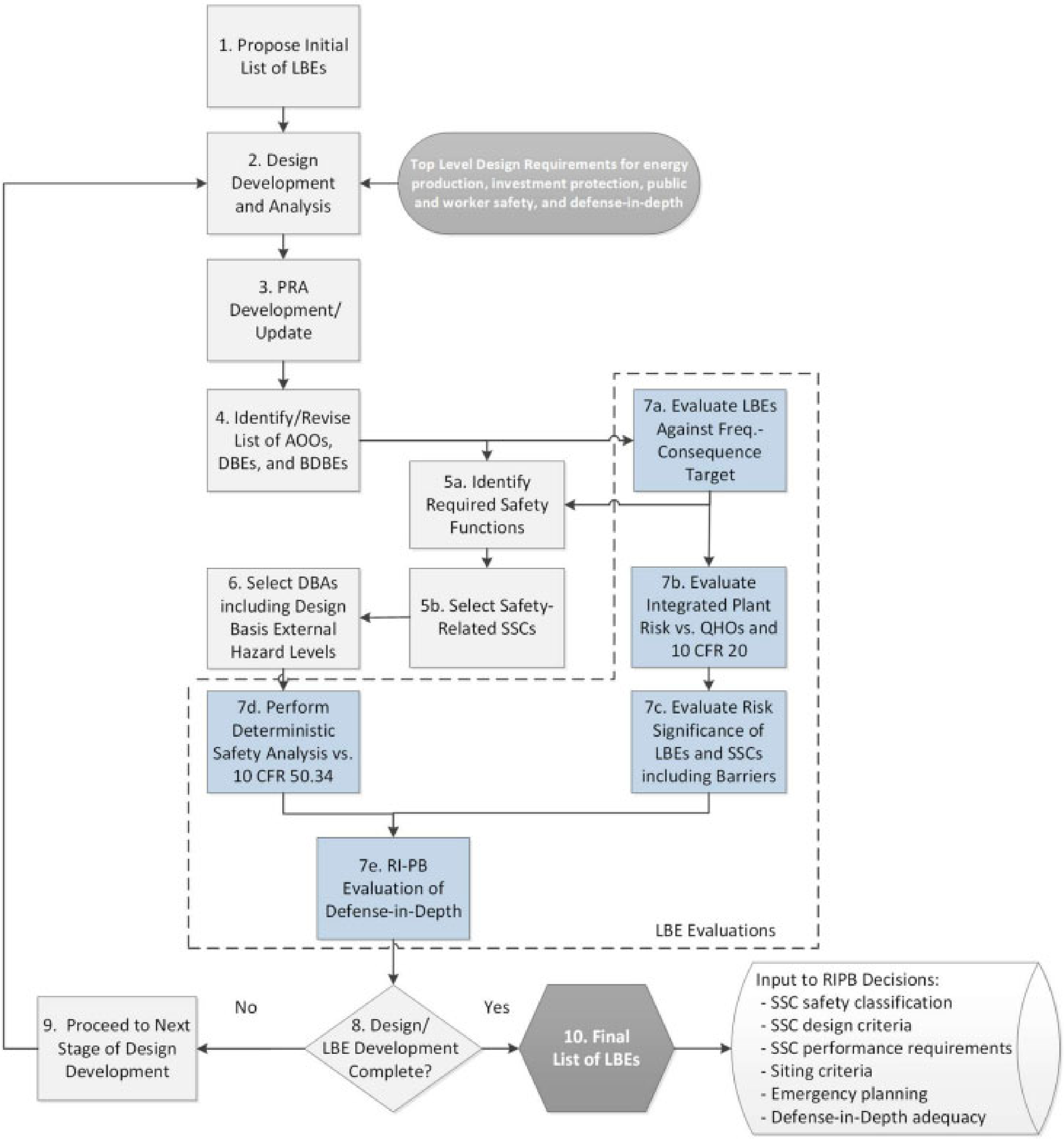

Figure 5: LMP process for selecting and evaluating LBEs [1]

3.4. Development of Required Safety Functions

3.4.1. Safety Functions in the MSRE OGS

Even without a comprehensive identification of LBEs for the inventory of radioactive material in the OGS, the results of the HAZOP study performed on the MSRE OGS, coupled with the results of the HAZOP study of the CCS (found in [17] and [18]]), can be used to help demonstrate how a designer could use the NEI 18-04 approach to determine the safety functions used in the prevention and mitigation of the OGS LBEs and the SSCs that perform these functions. As an initial step, Figure 6 displays a hierarchical representation of the safety functions for the MSRE, starting with the broad safety functions applicable to nuclear power plants in general (highest level) and ending with the specific safety functions employed by the MSRE OGS to prevent and/or mitigate LBEs (lowest level).

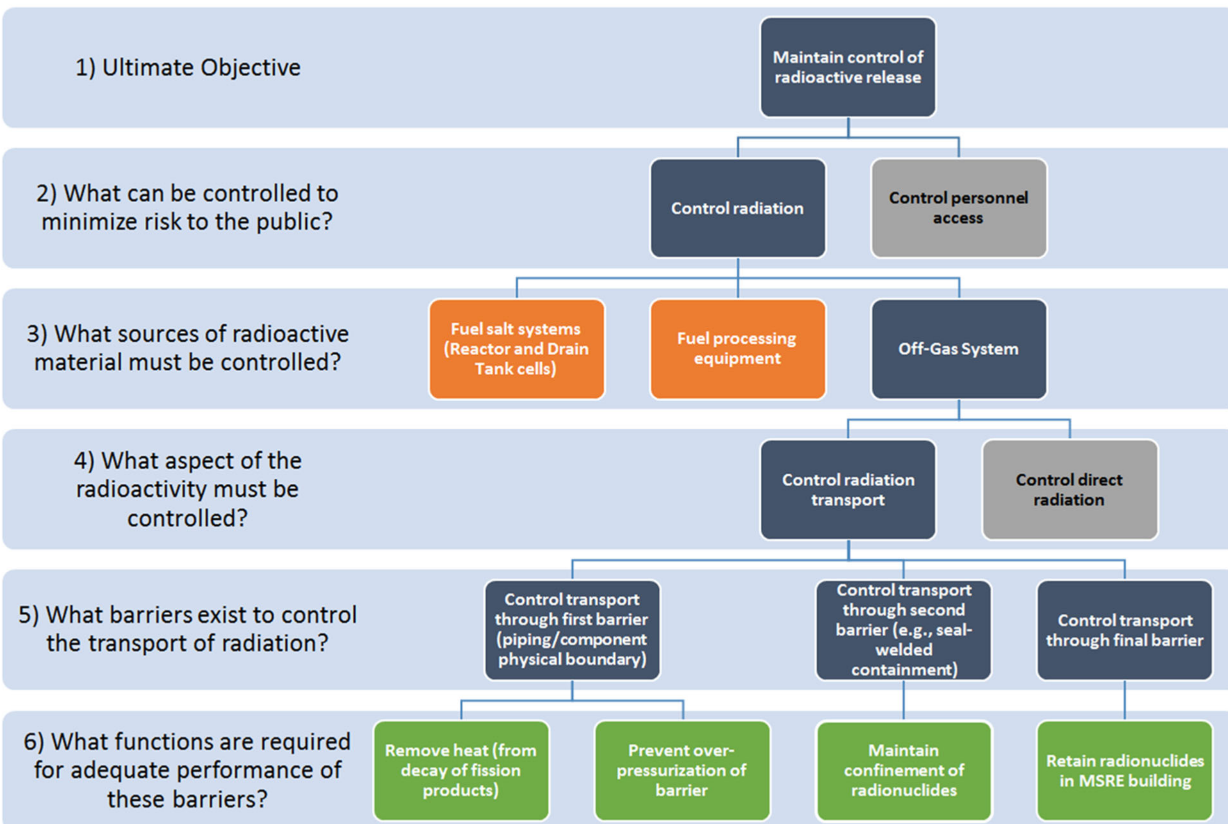

Figure 6: Decomposition of safety functions for MSRE OGS

Levels 1 and 2 of Figure 6 correspond to the highest levels of the safety function decompositions consistent with those defined for the X-energy reactor design [2] and the Modular High Temperature Gas Reactor (MHTGR) design [43]. The boxes in the third level of the tree represent the major sources of radionuclides in the MSRE design, discussed in detail in Ref. [16]. The fourth level of the tree distinguishes between the need to control the transport of radioactive material and to control direct radiation. Because the consequence of interest in this report is dose at the EAB, the functions supporting control of direct radiation are not further discussed.

The fifth level of Figure 6 divides the safety functions for the OGS based upon which barrier they are intended to protect. From the PHA results, and also evident from the above discussion of OGS-2, there are three distinct barriers to the release of radioactive material from the MSRE OGS. The first barrier is the physical boundary of the OGS piping and components. From the "Consequences" and "Safety System" columns of the HAZOP study results, it was determined that two phenomena could lead to transport of radioactive material through this barrier: excessive heat or excessive pressure. If radioactive material is released through this first barrier, the second barrier for this material is composed of several different smaller confinement structures in different locations around the MSRE building. [18] In the case of OGS-1, OGS-2, and OGS-3, this second barrier consists of the reactor cell and the CCS piping/components.[19]

The MSRE system is designed such that elevated levels of radioactivity in the CCS should trigger actions that isolate the cell exhaust flow to the MSRE stack without operator action. Finally, the last barrier to release of certain radioactive materials from the MSRE OGS is composed of the filters in the MSRE ventilation system. It is worth noting that this barrier would not be effective at confining all radionuclides, as noble gases would pass through the filters unmitigated. However, the "absolute" stack filters were intended to retain more than 99.9% of the particulates in the cell exhaust flow.[20]

<u>3.4.2 Exploring RSFs in the MSRE OGS</u>

Table 2 provides examples of SSCs that perform each OGS safety function and if it was able to be determined from the MSRE HAZOP study results, and preliminary ETA/FTA, whether the safety function was required to retain radionuclides within the F-C target. The list of safety functions and associated SSCs was created mostly based upon the results table from the MSRE HAZOP studies [17, 18], since the ETA/FTA has only quantified 3 LBEs to-date.

[18] See Section 3.2.2 of Ref. [16] for more detail on the barriers of the OGS.

[19] Although a HAZOP study has been conducted on the CCS, comprehensive PHA results are not available for the MSRE reactor cell.

[20] In Ref. [7], it is stated that the design basis of the HEPA filters is 99.97% efficiency for particles larger than 0.3 micrometers.

Table 2: Details of MSRE OGS Safety Functions

Safety Function	OGS Barrier	Example SSCs Performing Function	RSF? (based on OGS-2 and OGS-3)
Remove heat (from decay of fission products)	First barrier (components/piping)	Charcoal bed cooling, holdup volume cooling, particle trap cooling	Not investigated (see below)
Prevent over-pressure	First barrier (components/piping)	Standby charcoal bed, fuel salt pump bowl trips and alarms, rupture disks [in cover gas system],	Not able to be determined
Maintain confinement of radionuclides	Second barrier	Component cooling system, cell exhaust radiation alarms, cell exhaust isolation valve, charcoal beds, auxiliary charcoal bed	Not able to be directly determined, but suggested to be required
Retain radionuclides in MSRE building	Third barrier	Stack absolute filter, stack roughing filter, cell exhaust filter	Yes

Section 4 of NEI 18-04 states that required safety functions either:

1) Mitigate the consequences of DBEs to within the LBE F-C target and mitigate the consequences of Design Basis Events (DBAs, that only rely on the SR SSCs) to meet the dose limits of 10 CFR § 50.34 (2010) using conservative assumptions; or
2) Prevent the frequency of BDBEs with consequences greater than the 10 CFR § 50.34 dose limits from increasing into the DBE region (and beyond the F-C Target).

The only DBEs identified to-date are OGS-2 and OGS-3. Because both of these DBEs are pressure transients and not driven by the heat in the system due to fission product decay, the impact of not being able to remove this heat was not further evaluated in this Case Study. Additional LBEs involving the radioactive material in the OGS would need to be identified in order to fully evaluate this safety function.

In OGS-2, the SSCs performing the safety function of preventing overpressure impact the estimated frequency. However, if the standby charcoal bed is not included in the FT model for the initiating event, the number of cutsets is reduced by 8, but the estimated frequency of the sequences remain unchanged (and within the DBE region). Thus, the safety function of preventing overpressure in the OGS was not able to be shown to satisfy either of the criteria listed above and cannot be determined to be an RSF based on the LBEs currently identified. It is possible that this safety function could be shown to be an RSF using DBEs that have not yet been identified.

Similarly, based upon the definition of RSFs in NEI 18-04, the safety function of maintaining confinement of radionuclides was not able to be shown to be required. It is worth noting, however, that if the cell exhaust flow is not able to be isolated, there is no difference between OGS-1 and OGS-2; i.e., these event sequences would have the same consequences. This would seem to suggest that a DBA during which the function of maintaining confinement (by isolating cell exhaust flow) is NOT performed, the dose consequences at the EAB would exceed the 10 CFR 50.34 dose limit. Therefore, it seems that this safety function would be required.

The importance of retaining radionuclides within the third barrier to release (i.e., the MSRE building) was able to be demonstrated quantitatively using the preliminary MSRE analysis. To do so, calculations were performed to calculate the inhalation dose at the EAB due to the particulate fission products released during OGS-2 assuming the absence of any mitigating SSCs past the second barrier[21] [22]. The results showed an increase in inhaled dose that took the consequences of OGS-2 past the 10 CFR 50.34 dose limit. Thus, based on the LMP definition of RSFs, the function of retaining radionuclides in the MSRE building can be seen as being required. However, this conclusion is presented with the caveat that this a preliminary calculation. It is possible that a designer confronted with such results might not simply accept that this safety function would be required to mitigate the dose consequences, as described in NEI 18-04, such quantitative results could also suggest design changes that would decrease the risk associated with this event sequence.

3.5. Supporting the Classification of Safety-Related SSCs in the MSRE OGS

A comprehensive set of LBEs have not been modeled for the MSRE; therefore, the entire LMP approach for selecting SR-SCCs cannot be performed. However, because a DBE with relatively high consequences has been identified for the radioactive material in the MSRE OGS, the application of this process to OGS-2 will be investigated in this subsection.

Table 3 lists the SSCs and the functions that they perform as they are modeled in the MSRE ETA/FTA for OGS-2. It can be seen that the only functions that are modeled as successful in OGS-2 are the draining of the fuel salt from the fuel salt loop (corresponding to the safety function of maintaining confinement) and the retaining of radionuclides within the MSRE building.

[21] i.e., Assuming that 100% of the particulates are released via the MSRE stack.

Table 3: Details of MSRE OGS Safety Functions

SSC	Function	Function Success/Failure (OGS-2)	Prevention or Mitigation?
Piping (Line 522)	Withstand high pressure	Fail	Prevention
Particle trap	Prevent over-pressure	Fail	Prevention
OGS valves	Prevent over-pressure	Fail	Prevention
Charcoal beds	Prevent over-pressure	Fail	Prevention
Cell exhaust radiation monitors	Initiate isolation of cell exhaust flow and draining of fuel salt from fuel salt loop (maintain confinement)	Partial success (fuel salt drain)	Mitigation
Stack radiation monitors	Initiate isolation of cell exhaust flow (maintain confinement)	Fail	Mitigation
Cell exhaust isolation valve	Isolate cell exhaust flow (maintain confinement)	Fail	Mitigation
Freeze valve subsystem	Drain fuel salt from reactor (maintain confinement)	Success	Mitigation
Cell exhaust filter	Retain radionuclides in MSRE building	Success	Mitigation
Stack HEPA filters	Retain radionuclides in MSRE building	Success	Mitigation

Based on the successful functions in OGS-2, the first SSC that emerges as a candidate to be made safety-related is the freeze valve subsystem. Draining the fuel salt from the fuel salt loop during OGS-2 reduces the amount of time that the off-gas flow can leak into the reactor cell, which reduces the amount of radioactive material that is ultimately released via the stack. The results of the MSRE PHA studies conducted for this Case Study also suggest other events in which this particular safety function would need to be achieved by the freeze valve system (such as a rupture of tube in the heat exchanger). However, further analysis is needed to fully justify this decision, due to the possibility that the freeze valve subsystem may be needed to perform different RSFs for the inventory of radioactive material in the fuel salt loop (i.e., draining of the fuel salt may be needed for control of heat generation).

The “absolute” and “roughing” HEPA filters on the gas flow to the MSRE stack may also represent potential SR-SSCs. Although the small metallic filter in the cell exhaust line also performs the mitigation function of retaining radionuclides in the MSRE building during OGS-2, the results of the MSRE PHA suggest other events in the OGS that would not pass through the cell exhaust line. Based on the approach suggested in NEI 18-04, if the SSC is not available for all DBEs, it cannot be classified as SR. However, because the analysis above suggests that the safety function of retaining radionuclides in the MSRE building is required, it may make sense for the stack filters to be classified as SR. Additional analysis would be needed to confirm if these filters are available to perform the RSF for all applicable DBEs.

Although the available results suggest that the two example SSCs discussed above are candidates for SR-SSCs, no final conclusions can be made about other SSCs listed not being SR until all scenarios are analyzed.

3.6. Introduction to the LMP Process for Defense-in-Depth Adequacy Evaluations

Table 4 lists elements of DID that were identified for the MSRE OGS during the analysis of OGS-2. The information in this table is not intended to be a comprehensive list of all the provisions that contribute to DID for all LBEs that can occur involving the radioactive material in the OGS. Rather, the list is intended to illustrate the approach suggested in NEI 18-04 to evaluate LBEs using the layers of defense concept. The DID provisions in the table were able to be identified using the MSRE FTA/ETA, as well as the PHA results. The success criteria from NEI 18-04 are displayed in Figure 7 and were useful in determining the layer to which each provision belonged. It is important to note that some of the Plant Capabilities in Table 4 would also involve Programmatic aspects of DID. Additionally, because the MSRE design as analyzed has a TRL ~4-5, there was not much information available regarding Layer 5 of DID.

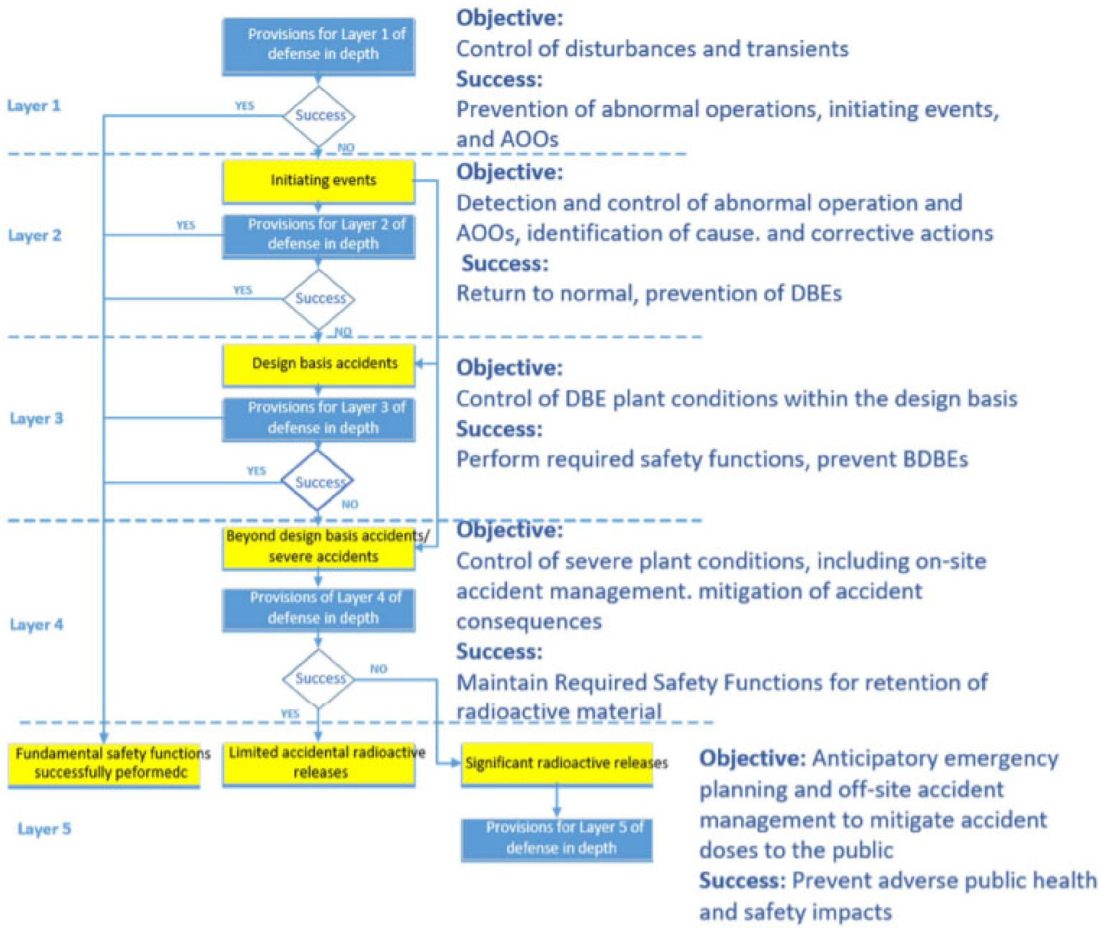

Figure 7: Framework for evaluating LBEs using layers of defense concept [1]

Table 4: Elements of DID identified for MSRE OGS during analysis of OGS-2

Provision	Objective	Plant Capability*/ Programmatic DID	Layer (from [1])
Robustness of OGS piping (Line 522)	Prevent leak of OGS flow	Plant Physical Capability	Layer 1
Design of OGS valves and filters	Prevent plugging in OGS	Plant Physical Capability	Layer 1
Valve lineups in OGS	Prevent unintended path for off-gas flow	Programmatic	Layer 1
Fuel salt pump bowl pressure indications	Notify operator to prevent over-pressurization in OGS	Plant Functional Capability	Layer 2
OGS pressure indications and alarms	Notify operator to prevent over-pressurization in OGS	Plant Functional Capability	Layer 2
Availability of standby charcoal bed	Allow for bypassing of plugged charcoal bed, prevent over-pressurization in OGS	Plant Functional Capability	Layer 2
Cell exhaust radiation indications and alarms	Notify operator to prevent release of radioactive material to atmosphere	Plant Functional Capability	Layer 3
Cell exhaust isolation (HCV-565-A1)	Isolate cell exhaust flow from MSRE stack (prevent release of radioactive material)	Plant Functional Capability	Layer 3
Fuel salt drain	Minimize duration of OGS leak from Line 522 (minimize release of radioactive material)	Plant Functional Capability	Layer 4
Stack radiation monitors	Notify operator of radioactive material release (minimize release of radioactive material)	Plant Functional Capability	Layer 4
ORNL monitoring of stack radiation monitors	Redundant means to notify operators in order to minimize duration of release of radioactive material	Programmatic	Layer 4
Cell exhaust filter	Retain radioactive material in cell exhaust flow to stack (minimize release of radioactive material)	Plant Functional Capability	Layer 4
Stack HEPA filters	Retain radioactive particles in flow to MSRE stack (minimize release of radioactive material)	Plant Functional Capability	Layer 4

*Note: It is possible that some of these plant capabilities would also involve programmatic aspects of DID

4. Conclusions and Observations

4.1. Conclusions

Quantitative estimates of frequency and consequence were developed for event sequences involving the radioactive material in the OGS of the MSRE. Significant and unquantifiable uncertainties were present in the dose consequence estimates; however, recognition of these uncertainties and the potential impacts they have on the resulting calculations would prompt an early designer to prioritize necessary analysis and tests aimed at improving the understanding of the source term. Using the LBEs identified, the application of the LMP approach was explored to illustrate how an advanced reactor developer at an early stage of design could initiate the first iteration of tasks such as identification of RSFs, classification of SR-SSCs, and evaluation of DID.

In spite of the uncertainties, limited quantitative results coupled with the qualitative results of PHA studies proved useful to provide an example of how to initiate these tasks, especially identification of safety functions and investigation of DID. Obviously, a more comprehensive and mature model of system risk would be required in order to make conclusions about LBEs and/or overall plant risk. The results of this analysis also suggest that, for advanced reactors that have distinct inventories of radionuclides, the NEI 18-04 process for selecting and evaluating LBEs (shown in Figure 5) will need to separately consider each inventory.

4.2. General Observations

A more detailed discussion of safety function decomposition in the LMP Guidance would be helpful. The process does not appear to be explained in enough detail; therefore, it was necessary to reference the original MHTGR safety function decomposition as well as the X-energy example in order to decompose the safety functions of the MSRE.

It is unclear if a PRA Safety Function (PSF) that prevents a DBE from having the frequency of an AOO would qualify as an RSF or if SSCs performing these safety functions would require any special treatment.

Discussion on the roles of the Integrated Decision-Making Panel (IDP) are dispersed amongst the document and do not seem to be collected in one place. For example, it is unclear if DID can be evaluated by parties other than the IDP.

The differentiation between “limited” and “significant” releases is not clear. The EPRI-VU team had questions regarding if 4 layers of DID are required and how it would be handled if a failure before Layer 4 could result in significant releases of radioactive material.

Based on the information provided in NEI 18-04, the following information flow is assumed: (1) LBE identification, (2) DID evaluation, (3) RSF identification, and (4) SSC classification. However, there is not a diagram or discussion in the current LMP guidance that outlines this flow of information.

4.3. Observations Related to Applying an LMP Approach to MSR Designs

NEI 18-04 currently lacks clarity on whether the LMP intends to provide guidance on consideration of non-radiological hazards. Event definitions provided in NEI 18-04 do not confine AOOs, DBEs, or BDBEs to just event sequences that could release radionuclides. For example, "The entire collection of event sequences considered in the design and licensing basis of the plant, which may include one or more reactor modules" could include event sequences with consequences related to chemical exposure, especially for MSRs. Additionally, it is stated that "LBEs may or may not involve release of radioactive material and may involve two or more reactor modules or radionuclide sources."[1]

In general, neither the LMP LBE identification and evaluation approach nor the SSC classification do not appear to exclude any considerations important for assessment of MSRs In particular, the SSC subsection on barriers[22] works particularly well for MSRs as it includes functional barriers. It is worth noting that there is an opportunity to add a sentence or two mentioning that SSC classification and DID needs to be performed for each inventory of radionuclides – which is important for non-LWR technologies (such as MSRs) that have multiple radioactive material inventories to be addressed.

The discussion in Section 5.7 of Ref. [1] does not necessarily preclude the "generalized" model from being used for inventories of radionuclides that are not the reactor fuel (e.g., OGS); however, a truly generalized model should mention the possibility of other types of radionuclide inventories -- rather than mentioning only the <u>possibility</u> of other barrier configurations. It may also be helpful to mention functional barriers during the conversation of barriers in this section, as the barriers represented in the example are all physical barriers (i.e., fuel barrier, coolant pressure boundary, and reactor building barrier).

4.4. Observations related to Applying LMP Approach to Early Stage Designs

Task 3 (Ref. [1], page 11) makes it clear that early stage safety evaluations will be different and have less detailed information than evaluations at later stages. Additionally, Tasks 1 through 3 describe the potential for using initial safety insights from early stage safety analyses to identify SR-SSCs and potential LBE; however, further discussion in NEI 18-04 appears to drop this theme entirely. These gaps lead to some confusion with respect to applicability of various assessments. For example:

- At what design stage does it make sense to quantitatively evaluate event frequency?
- At what design stage does it make sense to perform quantitative evaluations of consequence?
- At what design stage does it make sense to initially perform a preliminary DID evaluation?
- At what design stage does it make sense to initially identify SR SSCs?

[22] Subsection 4.4.4

The determination of RSFs (and to an extent, SR-SSCs) according to NEI 18-04 relies upon quantitative evaluation of event frequency and consequence. It is possible that the maturity of a given technology could be at such an early stage that uncertainty cannot be quantified. One example encountered in the MSRE Case Study team was the behavior of iodine in an MSR. Additional physical tests are needed to understand dose consequences of some event sequences involving radioiodine; however, it is not clear that preliminary determination of DID and associated SR-SSCs need be deferred.

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*Note: These letter reports are expected to be consolidated and published as individual appendices in an upcoming EPRI technical report.

Appendix A: Discussion of MSRE Case Study Calculations

Table A-1 displays the calculations for the radioactivity being swept by the helium cover gas from the gas space of the fuel salt pump bowl into the OGS each second. The radioelements were grouped based on their expected state of matter (solid, liquid, or gas) at 150°F, which was the maximum ambient temperature of the reactor cell during normal operations [7, 10]. For the noble gases (i.e., Kr and Xe), direct (external) radiation from a plume of airborne radioactivity is significant, while the dose commitment from the inhalation is not significant and is frequently excluded from consequence calculations (for example, see Ref. [44]). For particulates, the opposite is true; i.e., direct radiation from radioactive particles is not significant but dose commitment due to inhalation of these particulates is significant. Additionally, the ability to leak to the reactor cell, as well as the mitigating effects of the filters in the MSRE CCS and ventilation systems, are different for gases and particulates.

The values in Ref. [39] are given in disintegrations/min/inch of pipe. The pipe is 1-inch ID; thus, the volume of a 1-inch section is π * (0.5 in)^2 * 1 in = 0.785 in^3 = 0.0129 L. Given a flowrate of 4 L/min through the pipe, 4 / 0.013 = 308 sections/min = 5 section volumes per second. Finally, to get curies of isotope per second flowing through the pipe the following conversion was used:

(X dis/min/inch) / (2.22e12 dis/min/Ci) * (5 section volumes per second) = Y curies flowing into line per second.

Table A-1: Estimated radioactivity of OGS flow in Line 522

Isotope	Gas or Particulate	dis/min/in	Source	Ci per inch of pipe	Ci/sec	Bq/sec
H-3	Gas	n/a	[45] page 5 (60 curies/day)	n/a	6.94E-04	2.57E+07
Kr-87	Gas	1.3E+14	[39] Table 7.4	58.56	292.8	1.08E+13
Kr-88	Gas	1.2E+14	[39] Table 7.4	54.05	270.3	1.00E+13
Rb-88	Particulate* (Note 1)	1.8E+13	[39] Table 7.4	8.11	40.5	1.50E+12
Kr-89	Gas	2.2E+14	[39] Table 7.4	99.10	495.5	1.83E+13
Ru-89	Particulate	8.2E+13	[39] Table 7.4	36.94	184.7	6.83E+12
Kr-90	Gas	7.3E+13	[39] Table 7.4	32.88	164.4	6.08E+12
Nb-95	Particulate	9E+12	[39] Table 7.4	4.05	20.3	7.50E+11
Xe-135	Gas	1.3E+13	[39] Table 7.4	5.86	29.3	1.08E+12
Xe-135m	Gas	5E+13	[39] Table 7.4	22.52	112.6	4.17E+12
Xe-138	Gas	1E+14	[39] Table 7.4	45.05	225.2	8.33E+12
Cs-139	Particulate* (Note 1)	1E+13	[39] Table 7.4	4.50	22.5	8.33E+11
Xe-139	Gas	1E+13	[39] Table 7.4	4.50	22.5	8.33E+11
Xe-140	Gas	1E+12	[39] Table 7.4	0.45	2.3	8.33E+10
Sr-91	Particulate	2E+12	[39] pg 74	0.90	4.5	1.67E+11
Nb-97	Particulate	1E+14	[39] pg 74	45.05	225.2	8.33E+12
Mo-99	Particulate	3.3E+13	[39] pg 74	14.86	74.3	2.75E+12
Ru-105	Particulate	3.4E+12	[39] pg 74	1.53	7.7	2.83E+11
Rh-105	Particulate	8E+12	[39] pg 74	3.60	18.0	6.67E+11
Te-129m	Particulate	2E+12	[39] fig 7.9	0.90	4.5	1.67E+11
Te-131m	Particulate	7.5E+11	[39] pg 79	0.34	1.7	6.25E+10
I-131	Particulate* (Note 2)	5E+12	[39] pg 79	2.25	11.3	4.17E+11
I-132	Particulate* (Note 2)	1.2E+13	[39] pg 81	5.41	27.0	1.00E+12
TOTAL		**1.00E+15**		**451.4**	**2257**	**8.35E+13**
SUM	**Gases Only**	**7.17E+14**		**323.0**	**1615**	**5.98E+13**

*Note 1: Elements that were liquids at 150°F were assumed to behave as particulates

*Note 2: Although elemental iodine is gaseous at 150°F, the iodine in the OGS was assumed to be in a compound that would behave as a particulate.

The calculations supporting the estimate of inhaled dose at the EAB due to the particulate fission products released during OGS-2 is shown in Table A-2. It was assumed that particulate radionuclides could only leak from the fuel salt pump bowl into the reactor cell while there is contact between fuel salt and cover gas in the fuel salt pump bowl; i.e., once the fuel salt level drops below the fuel salt pump bowl, the leaking of radioactive particulates into the reactor cell ends because the helium sparging ceases. It is assumed that after the OGS leak it takes 10 minutes for the drain signal to be initiated, another 10 minutes for the freeze valve in the drain line to completely melt, and 5 minutes for the fuel salt to drain out of the fuel salt pump minutes for the freeze valve in the drain line to completely melt[23], and 5 minutes for the fuel salt to drain out of the fuel salt pump bowl[24].

For the fission products in the form of particulates, the dose at the EAB would be due to inhalation[25]. The activity leaked to the reactor cell for particulate fission products were estimated by assuming a leak of the entire off-gas flow (of the composition shown in Table A-1) to the reactor cell volume for 25 minutes. For I-131 and I-132, it was assumed that 50% of the iodine released to the reactor cell plated out on surfaces in the reactor cell or along the release path, consistent with the MSRE SAR [25, 41]. Based on calculations in the MSRE SAR, the HEPA filters leading to the MSRE stack have an efficiency of 99.9% for particles larger than 0.5 micrometers [25]. Thus, it was assumed that 0.1% of the fission product particulates released to the cell were subsequently released to the atmosphere via the MSRE stack. A formula from Ref. [25] was used to estimate the number of curies inhaled at the EAB (d=3000m from the MSRE building) based on the total activity released via the stack.[26]

Finally, dose coefficients for inhalation were found for each isotope in Appendix G of Ref. [46] and used to calculate the total inhalation dose for each isotope based on the amount of inhaled activity.

[23] Ref. [10] reports that it took 9-11 minutes for FV-103 to completely melt.

[24] Ref. [7] states that the fuel salt pump bowl has a volume of 6.1 cubic feet, with 2.0 cubic feet typically occupied by gas. Ref. [10] reports a maximum drain time of 40 min for the 70 cubic feet of fuel salt. Analysis assumes linear proportionality.

[25] For tritium, the dose at the EAB would also be due to inhalation. However, even assuming that the entire 4.86E-04 Ci/sec of tritium calculated to be in the off-gas flow is released from the stack for 1 hour, the resulting inhalation dose at the EAB is calculated to be less than 0.1 millirem; thus, tritium is not included in calculations.

[26] Note: n is the stability parameter and n=0.23 was used for the calculations based on the discussion in Appendix A of Ref. [25].

Table A-2: Calculations supporting the estimate of inhaled dose at the EAB due to particulate fission products during OGS-2

Isotope	Ci/sec	Leaked to Rx Cell	Q (curies)	Inhaled curies	Dose conversion factor	Total inhalation dose (rem)
Rb-88	40.5	6.08E+04	60.81	1.97E-04	1.60E-11	0.012
Ru-89	184.7	2.77E+05	277.03	8.95E-04	4.40E-11*	0.146
Nb-95	20.3	3.04E+04	30.41	9.83E-05	1.80E-09	0.655
Cs-139	22.5	3.38E+04	33.78	1.09E-04	1.40E-11*	0.006
Sr-91	4.5	6.76E+03	6.76	2.18E-05	4.10E-10	0.033
Nb-97	225.2	3.38E+05	337.84	1.09E-03	4.50E-11	0.182
Mo-99	74.3	1.11E+05	111.49	3.60E-04	9.90E-10	1.320
Ru-105	7.7	1.15E+04	11.49	3.71E-05	1.80E-10	0.025
Rh-105	18.0	2.70E+04	27.03	8.74E-05	8.20E-11	0.027
Te-129m	4.5	6.76E+03	6.76	2.18E-05	7.90E-09	0.638
Te-131m	1.7	2.53E+03	2.53	8.19E-06	9.10E-10	0.028
I-131	11.3	1.69E+04	8.45	2.73E-05	7.40E-09	0.747
I-132	27.0	4.05E+04	20.27	6.55E-05	1.10E-10	0.027
TOTAL	**642.2**	**9.63E+05**	**9.35E+02**	**3.02E-03**		**3.844**

*Note 1: dose rate coefficients were not available in Ref. [46] for Ru-89 or Cs-139 because the half-lives of these isotopes are less than 15 minutes. However, for estimation purposes, it was assumed that the dose rate coefficient for Ru-89 was similar to that of Ru-94 ($T_{1/2}$=51.8 min) and that the dose rate coefficient of Cs-139 was similar to that of Cs-130 ($T_{1/2}$=29.9 min)

Calculations were also performed to estimate the external dose due to the release of noble gases at the MSRE EAB; these are displayed in Table A-3. In order to estimate the release rate of the radioactivity from the reactor cell to the atmosphere via the cell exhaust line, the bypass line, and ultimately the stack, the activity of the noble gases in Table A-1 was multiplied by 1 hour to estimate the total activity leaked from the off-gas line to the reactor cell volume for the duration of the leak.[27] To get the concentration in the reactor cell atmosphere for each radioactive isotope, this total activity was divided

[27] The noble gases were assumed to leak for 1 hour instead of the 25 min assumed for the particulates to account for the migration of these gases from potential pockets in the fuel salt loop, including voids in the graphite moderator inside the MSRE vessel. Based on discussion in Ref. [47], this is likely a conservative assumption.

by a volume of 211,370 L.[28] In order to use the diffusion factor calculated in the MSRE safety documentation [25], a release rate for each isotope was obtained by assuming that 1 L of cell atmosphere was released via the stack every minute. This release rate was multiplied by the diffusion factor (χ) calculated in Ref. [25] for a stack release under inversion atmospheric conditions at a distance of 3000m (i.e., χ =4.90E-5). Dose rate coefficients were then obtained for each isotope from Ref. [46], and the external dose rate at the EAB was calculated in rem/hr for each isotope.

TableA-3: Calculations supporting the estimate of external dose at MSRE EAB for OGS-2

Isotope	Ci/sec	Leaked to Rx cell (curies)	Rx Cell Conc. (Ci/L)	Release rate (Ci/sec)	Act. Conc. In air (uCi/cc)	Dose Rate Coefficient (Sv/d per Bq/m3)	Dose Rate at EAB (rem/hr)
Kr-87	2.93E+02	1.05E+06	4.99E+00	8.31E-02	4.07E-06	3.40E-09	2.14E-03
Kr-88	2.70E+02	9.73E+05	4.60E+00	7.67E-02	3.76E-06	8.40E-09	4.87E-03
Kr-89	4.95E+02	1.78E+06	8.44E+00	1.41E-01	6.89E-06	[a]Note 1	2.89E-02
Kr-90	1.64E+02	5.92E+05	2.80E+00	4.67E-02	2.29E-06	4.98E-09[b]	1.76E-03
Xe-135	2.93E+01	1.05E+05	4.99E-01	8.31E-03	4.07E-07	9.60E-10	6.03E-05
Xe-135m	1.13E+02	4.05E+05	1.92E+00	3.20E-02	1.57E-06	1.60E-09	3.87E-04
Xe-138	2.25E+02	8.11E+05	3.84E+00	6.39E-02	3.13E-06	4.70E-09	2.27E-03
Xe-139	2.25E+01	8.11E+04	3.84E-01	6.39E-03	3.13E-07	4.70E-09[c]	2.27E-04
Xe-140	2.25E+00	8.11E+03	3.84E-02	6.39E-04	3.13E-08	4.70E-09[c]	2.27E-05
TOTAL	**1.61E+03**	**5.81E+06**	**2.75E+01**	**4.58E-01**	**2.25E-05**		**4.07E-02**

[a]Note 1: a dose coefficient of 4.2 mrem/hr per uCi/m^3 was used for Kr-89 [48]

[b]Note 2: a dose coefficient of 14.6 mrem/yr per uCi/m^3 was used for Kr-90 [49]

[c]Note 2: dose rate coefficients were not available in Ref. [46] for Xe-139 (decay mode: 100% 5.06 MeV β^-, $T_{1/2}$ = 39.68 s) or Xe-140 (decay mode: 100% 4.06 MeV β^-, $T_{1/2}$ = 13.6 s) because the half-lives of these isotopes are less than 15 minutes. However, for estimation purposes, it was assumed that the dose rate coefficients of these isotopes could be approximated using that of Xe-138 (decay mode: 100% 2.74 MeV β^-, $T_{1/2}$ = 14.08 min)

[28] The reactor cell was assumed to have a volume of π * (12 ft)^2 * 33 ft = 14,929 ft^3 = 422,742 L and half of this volume was assumed to be occupied by equipment (instead of atmosphere).